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PULSED-NEUTRON-SOURCE STUDIES WITH THE NASA ZERO POWER REACTOR II

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16. Abstract

An experimental and analytical study of the pulsed-source technique for measuring shutdown reactivities has been made using the NASA Zero Power Reactor II (ZPR-II) over a wide range of fuel solution concentrations for configurations varying from delayed critical to highly subcritical. The fuel is an aqueous solution of enriched (93.2 percent U^{235}) uranyl fluoride (UO_2F_2). Calculated fundamental prompt-mode decay constants obtained using one-dimensional multigroup S_n transport theory are compared to experimental values. Also, calculated values for the subcritical reactivity are compared with values derived from experiments using an inhour technique, the technique of Simmons and King, and the three well-known "area-ratio" techniques of Garelis and Russell, Gozani, and Sjöstrand.

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SUMMARY

An experimental and analytical study of the pulsed-source technique for measuring shutdown reactivities has been carried out in connection with the cylindrical NASA Zero Power Reactor II (ZPR-II). The ZPR-II, a solution system that can be used to assemble a wide range of unreflected, homogeneous, thermal reactors of simple geometry, has been used to test one-dimensional multigroup S_n transport theory calculations.

Pulsed-sourced experiments were carried out at four fuel concentrations having hydrogen-to-uranium-235 atom ratios (H/X) of 510, 975, 1503, and 1600. Data from each experiment have been reduced to give the fundamental prompt-mode decay constant and five separate evaluations of the shutdown reactivity. These results have been compared to values obtained from one-dimensional axial S_n calculations which used a buckling iteration technique to compute the radial leakage. The range of shutdown reactivity was greater than $25 for each concentration examined.

Of the five separate evaluations used for obtaining experimental shutdown reactivity, an inhour technique and the "area-ratio" technique of Gozani gave results in best agreement with the multigroup calculations. The "area-ratio" technique of Garelis and Russell gave results in slightly poorer overall agreement with the calculations. The technique of Simmons and King gave results that were in much poorer agreement with the calculations, especially at far shutdown reactivities, except for the H/X of 1600 systems where the agreement was very good. The method of Sjöstrand gave values having the poorest agreement with the calculations.

Although this work was confined to a single reactor tank, the range in bucklings was quite large, thus providing a severe test for the calculational model. The work also demonstrates the feasibility of using the pulsed-source experimental technique for a wide range of solution reactors.

INTRODUCTION

There are two basic methods for experimentally determining the reactivity of a subcritical multiplying medium that apply to the pulsed-source technique. One of the methods determines the reactivity from the prompt- and delayed-neutron responses and gives rise to the so-called "area-ratio" techniques. The experiments which use the delayed-neutron response as part of the technique for experimentally determining the subcritical reactivity of a system are also called modified pulsed-source experiments. Three area-ratio techniques for obtaining subcritical reactivities are considered in this study: (1) the area-ratio technique of Sjöstrand (ref. 1), (2) the extrapolated area-ratio technique of Gozani (ref. 2), and (3) the area-ratio technique of Garelis and Russell (ref. 3).

The other method for experimentally determining the reactivity of a subcritical multiplying medium makes use of the measured fundamental prompt-mode decay constant. Two techniques based on this method are considered in this study. In one technique the subcritical reactivity is obtained by using the measured fundamental prompt-mode decay constant and a calculated parameter. This technique is referred to as the inhour method in this discussion and is described by Preskitt, Nephew, Brown, and Van Howe (ref. 4). The other technique, that of Simmons and King (ref. 5), uses only the fundamental prompt-mode decay constants at the subcriticality in question and also at delayed critical.

Modified pulsed-neutron-source techniques for experimentally determining the reactivity of a subcritical multiplying medium have been based on simplified reactor models. The area-ratio method of Sjöstrand is based on monoenergetic diffusion theory applied to an unreflected reactor; the extrapolated area-ratio method of Gozani is based on the assumption that the spatial variation of the delayed modes is the same as that of the fundamental prompt mode; and the technique of Garelis and Russell is also based on monoenergetic diffusion theory applied to an unreflected reactor. However, these experimental techniques are used for complex reactor systems which require correspondingly more complex reactor models. Attempts to improve the theoretical basis of the modified pulsed-neutron-source techniques have been made by Corngold (ref. 6), Becker and Quisenberry (ref. 7), Masters and Cady (ref. 8), Wallace, Teare, and Greene (ref. 9), and Preskitt, Nephew, Brown, and Van Howe (ref. 4).

The difficulties associated with the various modified pulsed-neutron-source techniques, as well as with the method of Simmons and King, indicate the need for data obtained for systems having a simple geometry. The NASA zero-power solution reactors are convenient for performing pulsed-source experiments and making corresponding calculations for systems ranging from delayed critical to highly subcritical. These reactors use a solution of enriched (93.2 percent U^{235}) uranyl fluoride salt (UO_2F_2) dissolved in water and contained in a cylindrical aluminum vessel. For a given concentration of uranyl fluoride salt in water, criticality is achieved by adjusting the height of the

solution. No control rods are associated with these solution reactors. The geometry is thus ideal for performing pulsed-neutron experiments. The reactor used for the pulsed-neutron experiments reported herein is described by Fox, Mueller, Ford, and Alger (ref. 10) and is officially designated as the NASA Zero Power Reactor II (ZPR-II).

Pulsed-source experiments performed with the ZPR-II at a single fuel concentration have been reported and described in detail in reference 11. The agreement between calculated and experimental results was generally close for systems up to \$5 shutdown but data were obtained as far as \$50 shutdown. Since the bare solution system presented an excellent opportunity to do pulsed-source experiments over a wide range of clean simple geometries and to use the data to test the calculational methods, the work of reference 11 has been extended. This report presents the experimentally derived data for four fuel concentrations and compares these data with the calculational results.

PULSED-SOURCE EXPERIMENTS

Description of Experimental Arrangement

The experiments carried out in this study were an extension of the work described in detail in reference 11. The typical pulsed-source geometry for an unreflected solution reactor is shown in an elevation view in figure 1. The pulsed source and neutron detector were located at positions determined in reference 11 to be near the optimum. The

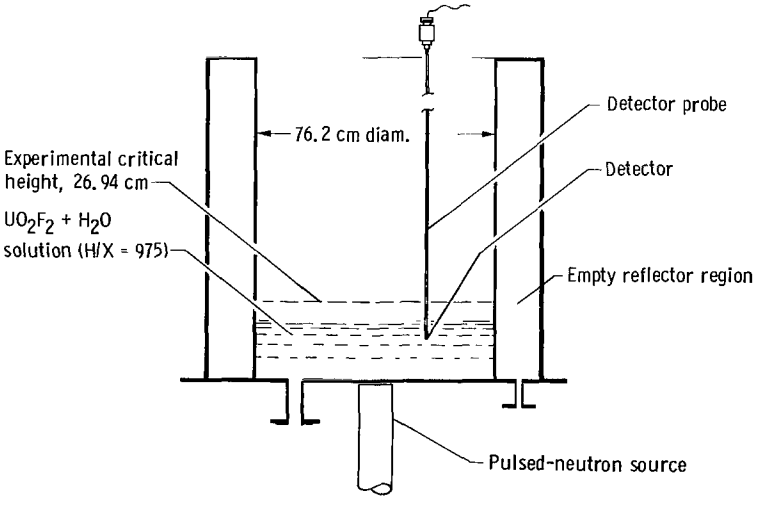


Figure 1. - Typical pulsed-source geometry for unreflected NASA solution reactor. (Drawn to scale.)

source was located on the axis of the 76.2-centimeter-diameter cylindrical core immediately under the bottom. The neutron detector, a tiny flux probe with a 1/v energy response, was located at a point about 17 centimeters radially displaced from the axis and two-thirds of the way up between the bottom and top surface of the fuel solution. Signal pulses from the neutron detector were sorted according to time in a 400-channel analyzer.

The general pulsed-source experiment is shown in the block diagram in figure 2.

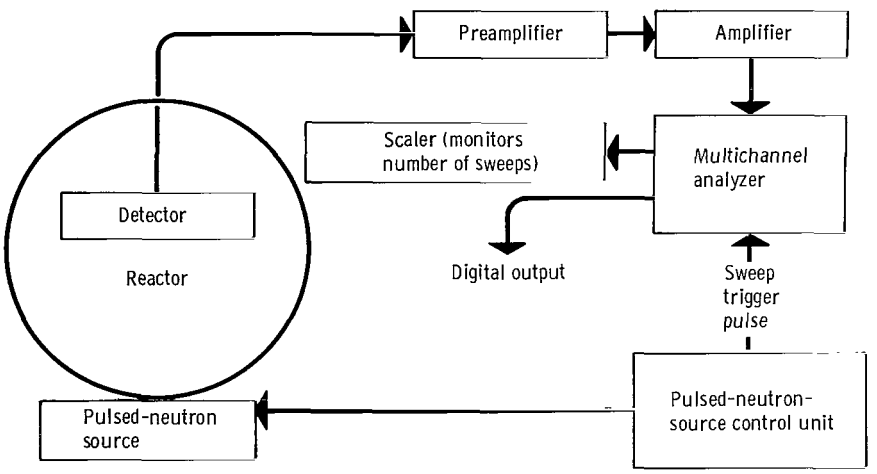


Figure 2. - Diagram of pulsed-neutron-source experiments in NASA solution reactor.

The experimental arrangement was such that each pulse of the neutron generator results in one time sweep of the analyzer. Pulse rate of the neutron generator was kept at 5 to 6 pulses per second and all subcritical experiments were prepulsed for about 5 minutes. The analyzer channel width was varied from 50 to 400 microseconds depending on the particular experimental configuration being studied. In this mode of use the analyzer had a fixed 12.5-microsecond time gap between channels. All 400 channels were always used with the channel width chosen to provide the best delayed-neutron counting statistics commensurate with enough channels (>25) from which to determine the prompt-neutron response.

Experimental Procedures

Experiments were performed at four fuel concentrations having hydrogen-to-uranium-235 atom ratios (H/X) of 510, 975, 1503, and 1600. The experimental critical heights for these fuel concentrations are about 18, 27, 56, and 72 centimeters, respectively, for the

unreflected 76.2-centimeter-diameter core. Results for the H/X of 975 were previously reported in reference 11 and are repeated here for completeness. The experiments included many core geometries covering a reactivity range from delayed critical to at least $25 shutdown for all concentrations.

For the pulsed-source experiments at delayed critical, the neutron level increases constantly with pulsing time so that initial level, pulsing time, and pulse size are controlled carefully to prevent the neutron level from exceeding the operating range of the neutron counting equipment or the deadtime counting losses from becoming excessive. The technique used for pulsing at delayed critical was to start at as low a neutron level as possible and, using small neutron bursts, to pulse for only a 5-minute duration.

The subcritical configurations were achieved by reducing the solution height below delayed critical. Since solution height can be varied continuously, a nearly unlimited number of configurations is possible. It is possible to reproduce fuel solution heights to within a few thousandths of a centimeter as a routine matter.

More configurations were pulsed near delayed critical to about $5 shutdown than at far shutdown ($25 to $50). Operating times can be much shorter for near critical configurations because of counting statistics limitations. The region up to about $5 shutdown represents the region of most practical value and interest. Most safety rods, control devices, experimental insertions, or other reactivity changes that are likely to be deliberately inserted into a critical system fall into this category. Experiments beyond $5 shutdown are of value for testing various calculational models and have use in connection with the experimental determination of the total shutdown worth of a reactor control system.

Reduction of Experimental Data

The reduction of the experimental data into the parameters of interest used the computer program GRIPE II developed by Kaufman (ref. 12) and adapted for use on the IBM-7094. This program calculated the fundamental prompt-mode decay constant as well as the reactivity by the "area-ratio" techniques developed by Gozani, Garelis and Russell, and Sjöstrand as discussed later. An error analysis of each parameter is also carried out by this computer program as well as a quality check of the raw data.

After the experimental data have been ordered according to time, they are corrected for counting losses. The reactor background was introduced as a constant for all channels. The experimenter makes an estimate of the number of data channels over which delayed background appears to be constant. The computer program then makes a statistical search to find the best value of the delayed background. The average value of total reactor plus delayed-neutron background is then subtracted from each data point thus

giving the prompt-neutron data. These prompt-neutron data are next examined by the computer program to find the region of best fit to a negative exponential function. This exponential is assumed to be the fundamental prompt mode. The best fit is determined by a least-squares analysis of the experimental data in which the data are weighted in importance approximately according to the number of counts per data channel. The channels which determine the fundamental prompt-mode region of the decay curve are determined by an iterative calculation which successively rejects channels from each end of the exponential portion of the curve until the best fit is obtained. The initial iteration is chosen to be conservatively large to assure that the best fit is clearly delineated. A χ^2 goodness-of-fit test is then applied to assure that the data are described by an exponential.

The reactivity values are determined for all iterations by the computer program. The reactivity values reported are for the iteration with the best decay curve fit. The error analysis is based on counting statistics only and is not indicative of the absolute accuracy of the experimental data.

Analysis of Experimental Data

Figure 3 is an idealized representation of the data obtained from a pulsed-source

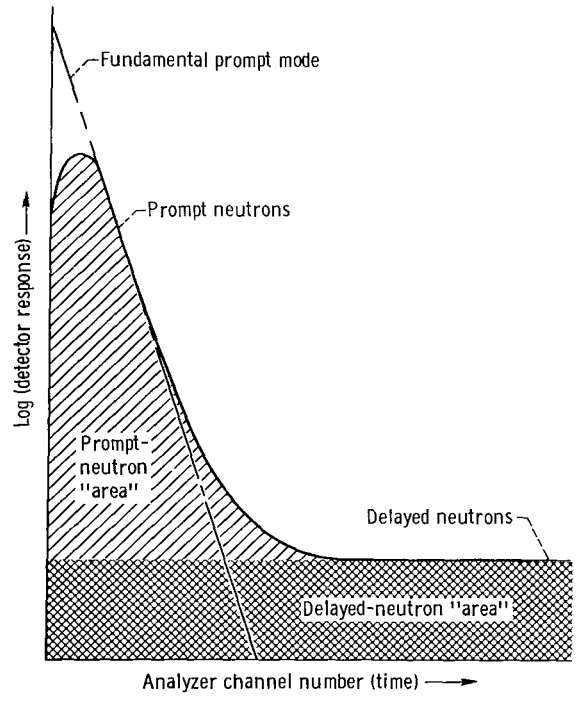


Figure 3. - Ideal detector response to pulsed-neutron source.

experiment. It is typical of the results to be expected from repetitively pulsing a subcritical configuration. The background-corrected logarithm of the detector response is plotted as a function of the analyzer channel number. The equilibrium delayed-neutron contribution persists after the decay of the prompt-neutron portion of the curve. The slope of the prompt portion of the curve yields the fundamental prompt-mode decay constant $\left(\alpha_o^p\right)_E$ where the subscript E denotes experimental. The areas corresponding to the time integrals of the prompt and delayed neutrons are also shown in figure 3. It is the time integral of delayed neutrons that is the major factor in determining minimum operating times with the far subcritical experiments. At least 1000 counts from delayed neutrons (no less than two counts per channel) were found to be necessary for desirable statistics when determining the delayed-neutron area or time integral.

The manner in which the experimental data are related to reactivity is indicated in this section. There are five techniques used in this report to determine shutdown reactivity from experimental data. Three of the techniques use the so-called area-ratio approach and the others make use of the fundamental prompt-mode decay constant in some fashion. The derivations of the various techniques can be found in appendix B of reference 11. The symbols used in this report are identical to those in reference 11.

Sjöstrand technique. - The reactivity in dollars $\rho_s^{SO}(\$)$ for the area-ratio technique of Sjöstrand is obtained by taking the ratio of the prompt-neutron area to the delayed-neutron area; that is,

$$\rho_s^{SO}(\$) = \frac{-\int_0^\infty N_p(\vec{r},t)\,dt}{\frac{N_d(\vec{r},t)}{R}}\, Z_o^{SO}(\vec{r})$$

where $N_p(\vec{r},t)$ is the total number of prompt neutrons per unit volume and unit time at position $\vec{r}$ and at time t; $N_d(\vec{r},t)$ is the total number of delayed neutrons per unit volume and unit time at position $\vec{r}$ and time t; R is the pulse repetition rate; and $Z_o^{SO}(\vec{r})$ is the spatially dependent distortion factor. The kinetic distortion factor for any of the area-ratio techniques is a quantitative measure of the spatial and energy differences between the delayed- and prompt-neutron modes. In this work these factors were taken as unity for all experiments. The denominator of this equation is obtained from the integral

$$\int_0^\infty N_d(\vec{r},t)\,dt$$

where the delayed-neutron distribution is assumed to be constant with time.

Gozani technique. - The reactivity in dollars for the area-ratio technique of Gozani $\rho_s^{GO}(\$)$ is obtained by taking the ratio of the extrapolated fundamental prompt-mode area to the delayed-neutron area; that is,

$$\rho_s^{GO}(\$) = \frac{\dfrac{Q_o N_o^p(\vec{r})}{\left(\alpha_o^p\right)_E}}{\dfrac{N_d(\vec{r},t)}{R}} Z_o^{GO}(\vec{r})$$

where Q_o is a coefficient for the fundamental prompt mode, $N_o^p(\vec{r})$ is the total number of neutrons of the fundamental prompt mode per unit volume at position $\vec{r}$, and $Z_o^{GO}(r)$ is the spatially dependent kinetic distortion factor. The numerator of this equation is obtained experimentally by fitting the equation

$$N_o^p(r,t) = Q_o \exp\left[\left(\alpha_o^p\right)_E t\right] N_o^p(\vec{r})$$

and integrating with time. The term $Q_o N_o^p(r)$ corresponds to the zero time intercept and $\left(\alpha_o^p\right)_E$ the slope of the curve in figure 3. The denominators of the two reactivity equations are identical.

Garelis and Russell technique. - In the Garelis and Russell technique of determining the reactivity $\rho_s^{GR}(\$)$ the prompt-neutron response is weighted by the factor exp (xt) and then integrated over time. The quantity x is adjusted so that this integral equals the prompt- plus the delayed-neutron area; that is,

$$\int_0^\infty N_p(\vec{r},t) \exp(xt)\, dt = \int_0^\infty N_p(\vec{r},t)\, dt + \frac{N_d(\vec{r},t)}{R}$$

where all symbols are as defined previously. The reactivity in dollars is then given by

$$\rho_s^{GR}(\$) = \left[\frac{\left(\alpha_o^p\right)_E}{x} + 1\right] Z_o^{GR}(\vec{r})$$

where $Z_o^{GR}(\vec{r})$ is the spatially dependent kinetic distortion factor and is equal to $Z_o^{GO}(\vec{r})$. The quantity x in the Garelis and Russell technique thus corresponds to the parameter

$\overline{\beta}_o/\Lambda_o$, the ratio of the effective delayed-neutron fraction to the generation time for the fundamental prompt mode.

Inhour technique. - Reactivity may also be determined from an inhour technique $\rho_s^{IN}(\$)$ by using the fundamental prompt-mode decay constant $\left(\alpha_o^p\right)_E$ and a calculated value for the parameter $\overline{\beta}_o/\Lambda_o$, where $\overline{\beta}_o$ is the total effective delayed-neutron fraction for the fundamental prompt mode and Λ_o is the generation time for the fundamental prompt mode. This reactivity is given by

$$\rho_s^{IN}(\$) = \frac{\left(\alpha_o^p\right)_E}{\frac{\overline{\beta}_o}{\Lambda_o}} + 1$$

Simmons and King technique. - The experimental reactivity may also be determined by the technique of Simmons and King. In this technique the fundamental prompt-mode decay constant must be measured at delayed critical and at the subcritical configuration in question. Then the reactivity $\rho_s^{SK}(\$)$ is given by

$$\rho_s^{SK}(\$) = -\frac{\left(\alpha_o^p\right)_E}{\left(\alpha_o^p\right)_E^{DC}} + 1$$

where the superscript DC denotes delayed critical.

MULTIGROUP TRANSPORT CALCULATIONS

The basic S_n transport calculation is discussed fully in appendix B of reference 11. Briefly, one-dimensional calculations to obtain both the fundamental prompt-mode flux and the static adjoint flux were carried out along the axis of these cylindrical reactor systems. These calculations then yield the calculated reactivity $\rho_s^C(\$)$ by using either of the following two equivalent expressions:

$$\rho_s^C(\$) = \frac{K_{eff} - 1}{K_{eff}\overline{\beta}_o}$$

or

$$\rho_s^C(\$) = \frac{\alpha_o^p}{\frac{\overline{\beta}_o}{\Lambda_o}} + 1$$

where K_{eff} is the neutron multiplication constant, α_o^p is the calculated value of the fundamental prompt-mode decay constant, and all other symbols are as defined previously.

The S_4 approximation to the angular flux and P_1 approximation to elastic neutron scattering were used to treat the neutron leakage from the ends of these subcritical solution reactors. Radial leakage was calculated by using a buckling iteration technique. Eight neutron energy groups were used, seven fast and one thermal group which included an up-scattering transfer component. The fast group cross sections were obtained using the GAM-II code (ref. 13) while the thermal cross sections were obtained using the GATHER-II code (ref. 14). The energy boundaries for each neutron energy group and the delayed-neutron data used in these calculations are the same as were used in reference 11. The neutron energy group split is given in table I, while the delayed-neutron data for uranium-235 are given in table II (from ref. 15).

A number of the parameters in the basic one-dimensional axial S_n transport calculations were varied in order to determine their effect on other parameters such as the fundamental prompt-mode decay constant for a given shutdown solution height. Many of the parameters proved to be factors of secondary importance. Although clean simple reactor geometries were used throughout this study, the major factor was how to account for radial leakage. A buckling iteration technique was used to compute the radial leakage

TABLE I. - NEUTRON ENERGY GROUP SPLIT

Group	Energy	Lethargy
1	14.92 MeV to 2.231 MeV	-0.4 to 1.5
2	2.231 MeV to 0.821 MeV	1.5 to 2.5
3	0.821 MeV to 0.224 MeV	2.5 to 3.8
4	0.224 MeV to 9.12 keV	3.8 to 7.0
5	9.12 keV to 454 eV	7.0 to 10.0
6	454 eV to 8.32 eV	10.0 to 14.0
7	8.32 eV to 0.414 eV	14.0 to 17.0
8	0.414 eV to 0 eV	17.0 to ∞

TABLE II. - DELAYED-NEUTRON DATA FOR URANIUM-235

[From ref. 15, p. 90; $\beta = 0.0065$.]

Delayed group	Decay constant, λ, sec^{-1}	Relative abundance, β_i/β	Delayed-neutron fraction, β_i
1	0.0124	0.033	0.000 214 5
2	.0305	.219	.001 423 5
3	.111	.196	.001 274 0
4	.301	.395	.002 567 5
5	1.14	.115	.000 747 5
6	3.01	.042	.000 273 0

for the range of fuel concentrations and fundamental prompt-mode decay constants that had been measured experimentally. The wide range of experiments provided a thorough test for this approximation. Two-dimensional calculations would provide a more accurate representation of the geometry and thus avoid the problem of how to represent the radial leakage in a one-dimensional analysis. However, such calculations were not made because a suitable two-dimensional transport program was not available.

The buckling iteration technique used a series of one-dimensional spatial solutions first in the axial and then the radial direction on alternate calculations. The transverse dimension was iteratively adjusted with each calculation until the identical value for K_{eff} was achieved for both the axial and radial solutions. The agreement between experimental and calculated critical heights was within a few percent in all cases as shown in table III. The radial dimension obtained in the final iteration is called the effective radius in this report and is always somewhat larger than the actual reactor radius. Note that the effective radius does not include the extrapolation length which is generated by the calculation and is different for each energy group.

TABLE III. - TABULATION OF SEVERAL KEY PARAMETERS AT DELAYED CRITICAL AS A FUNCTION OF FUEL SOLUTION CONCENTRATION

Hydrogen-to-uranium-235 atom ratio, H/X	510	975	1503	1600
Experimental critical height, cm	18.13	26.94	55.9	71.5
Calculated critical height, cm	17.95	26.32	54.72	73.5
Effective core radius,[a] cm	43.609	42.601	41.468	41.047
Experimental prompt-mode decay constant, $\left(\alpha_o^p\right)_E$, sec^{-1}	-192	-116	-80	-79.5
Calculated prompt-mode decay constant, α_o^p, sec^{-1}	-194	-116	-80.5	-77
Neutron generation time, Λ_o, μsec	39.618	64.560	86.277	89.759
Effective delayed-neutron fraction, $\bar{\beta}_o$	0.007792	0.007406	0.006999	0.006924
$\bar{\beta}_o/\Lambda_o$, sec^{-1}	196.68	114.71	81.119	77.14

[a]Effective core radius does not include extrapolation length.

The variation of the effective radius with solution height is presented for the four solution concentrations in figure 4. The solid line is used to indicate the values at delayed critical. Included in the figure are values of the effective radius for intermediate and for shutdown cases. These do not differ by more than 0.3 centimeter from the curve representing the values obtained at delayed critical. The effective radius as determined for the delayed critical case at a given concentration can be used for all cases at that concentration with little error. Calculated solution heights obtained using the effective radius obtained at delayed critical differed very little from fuel heights obtained by an independent buckling iteration calculation of the same case.

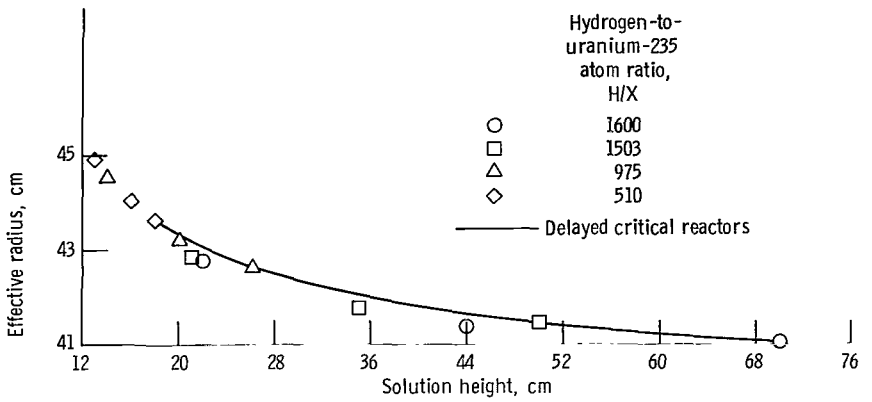


Figure 4. - Calculated effective reactor radius as a function of solution height for various fuel solution concentrations using buckling iteration technique.

Several parameters were varied for a solution reactor with an H/X of 975 to determine their effect on other parameters such as the fundamental prompt-mode decay constant. In one case the number of thermal energy groups was increased from one to four giving seven fast and four thermal energy groups. In the second case, the S_n quadrature and elastic scattering orders were increased to give an $S_{16}P_3$ model. Both of these more rigorous calculations involved much more time for computation. Calculations were made for the system at delayed critical and at about \$50 shutdown to test the effect of the higher order approximations. The calculated results of α_o^p differed by less than 0.5 percent between the more rigorous and simpler models.

The variation of the calculated parameters associated with the buckling iteration technique calculations are presented as a function of the ratio of shutdown fuel solution height to the height at delayed critical h/h_c in figures 5 to 8. Each figure presents the calculated data for one parameter at all four fuel concentrations. The range of calculated variables corresponds closely to the range of experimental investigation with one difference; the calculated parameters extend beyond delayed critical to prompt critical.

Figure 5 shows the calculated values of the effective neutron multiplication factor K_{eff}. Figure 6 shows the variation in the fundamental prompt-mode effective delayed-neutron fraction $\overline{\beta}_o$ where the value of $\overline{\beta}$ was taken as 0.0065 for uranium-235. Figure 7 shows the fundamental prompt-mode neutron generation time Λ_o which varies over a total range from about 40 to 113 microseconds. The parameter $\overline{\beta}_o/\Lambda_o$, the ratio of the parameters given individually in figures 6 and 7, is plotted in figure 8 also as a function of h/h_c. It is noted that $\overline{\beta}_o/\Lambda_o$ does not vary much at an H/X of 1503 and is nearly constant for h/h_c varying from 0.65 to 1.0 for H/X of 1600. The Simmons and King technique of experimentally determining shutdown reactivity assumes that $\overline{\beta}_o/\Lambda_o$ is a constant and therefore should give good results at H/X of 1600.

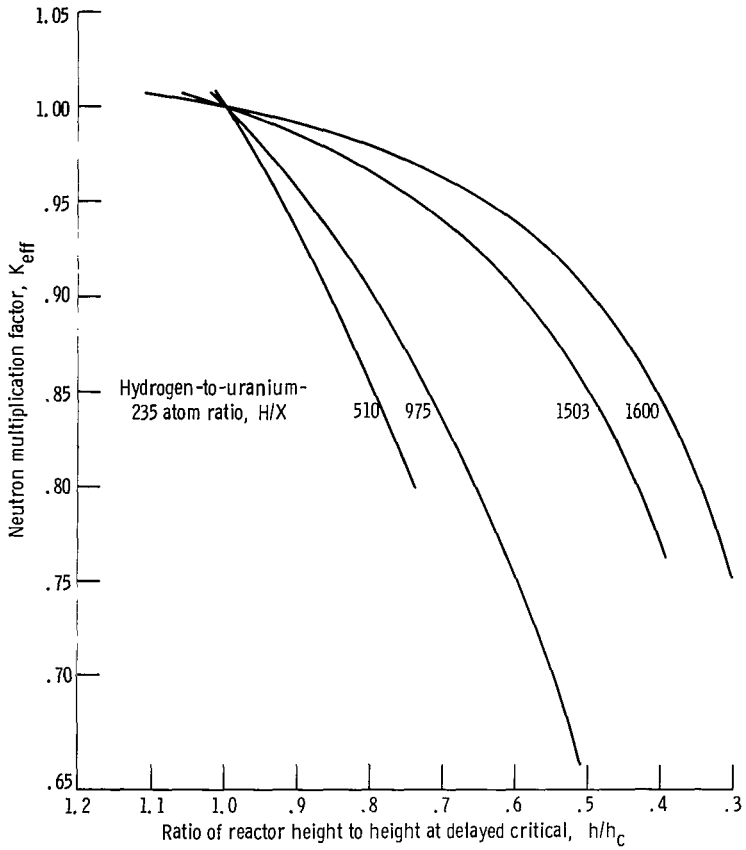


Figure 5. - Neutron multiplication factor as a function of fuel solution height for various fuel concentrations. (See table III for solution heights at delayed critical.)

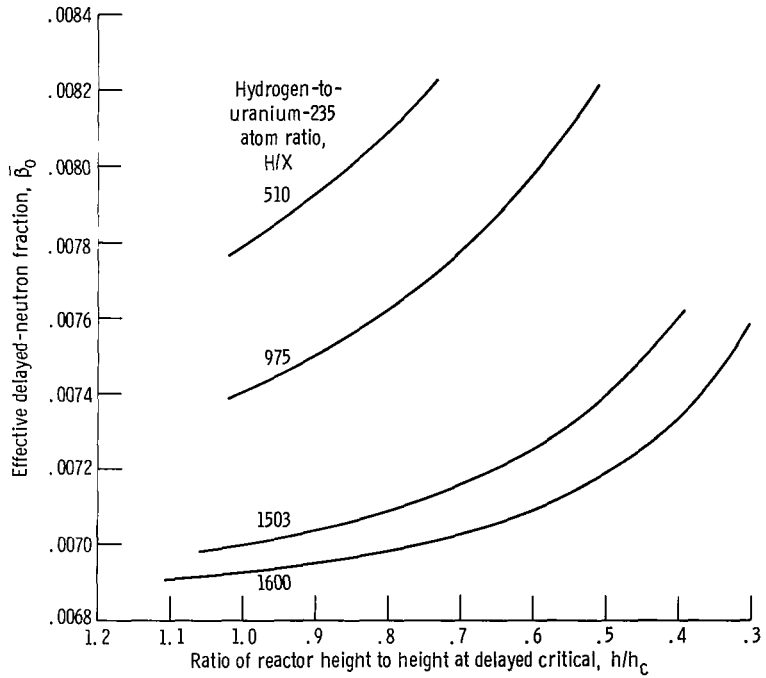


Figure 6. - Effective delayed-neutron fraction as a function of fuel solution height for various fuel concentrations. (See table III for solution heights at delayed critical.)

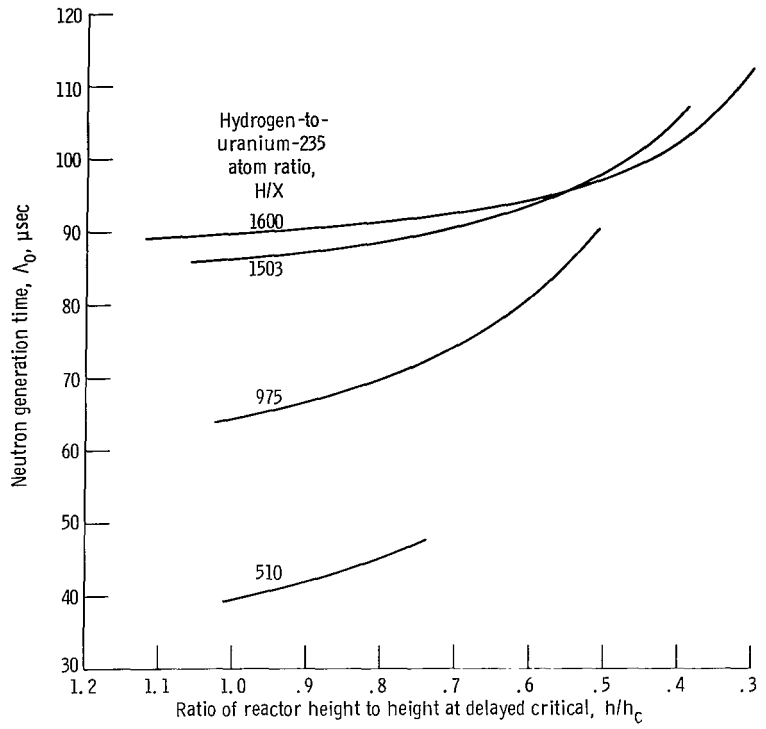


Figure 7. - Neutron generation time as a function fuel solution height for various fuel concentrations. (See table III for solution heights at delayed critical.)

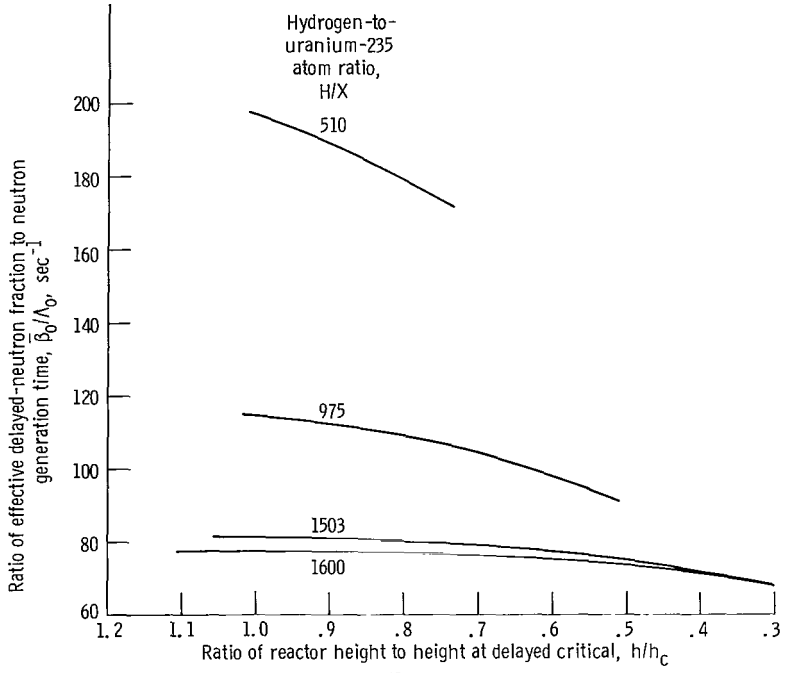


Figure 8. - Calculated parameter $\bar{\beta}_0/\Lambda_0$ as a function of fuel solution height for various fuel concentrations. (See table III for solution heights at delayed critical.)

COMPARISON OF EXPERIMENTS AND CALCULATIONS

Calculations and experiments have been performed at four fuel concentrations (H/X values of 510, 975, 1503, and 1600) and for a wide range of fundamental prompt-mode decay constants at each concentration for the NASA solution reactor ZPR-II. The calculated and experimental values are compared in several ways in this section. The experimental results obtained from the pulsed-source study are given as two parameters: (1) the fundamental prompt-mode decay constant, and (2) shutdown reactivity values using several techniques for analyzing the experimental data. The experimental data are compared in each case with a calculated value of the fundamental prompt-mode decay constant and of the shutdown reactivity.

The values at delayed critical of several key parameters are tabulated in brief form in table III. Experimentally measured heights at delayed critical can be compared with calculated heights for all values of H/X studied. The effective core radius obtained in calculating the height at delayed critical is also tabulated. Experimental and calculated values at delayed critical of the fundamental prompt-mode decay constant can be compared as well as the fundamental prompt-mode calculated parameters of neutron generation time Λ_o, the effective delayed-neutron fraction $\bar{\beta}_o$, and the ratio parameter $\bar{\beta}_o/\Lambda_o$. It is noted that $\alpha_o^p = -\bar{\beta}_o/\Lambda_o$ at delayed critical. The absolute values of α_o^p and $\bar{\beta}_o/\Lambda_o$ at delayed critical agree within a few percent with the best agreement occurring for the dilute concentrations.

All data for the fundamental prompt-mode decay constant are presented in table IV, and all shutdown reactivity data are presented in table V. The experimental data have been corrected for temperature (to 20^o C) and for water evaporation from the fuel solution. The temperature correction is made by using the experimentally determined temperature coefficients reported in reference 10. The correction for evaporation is carried out using the relation described in appendix B of reference 16. Both corrections were small since the variation in fuel temperature and concentration was kept small during the experiments.

Fundamental Prompt-Mode Decay Constants

A comparison of experimental and calculated values of the fundamental prompt-mode decay constant α_o^p in seconds^{-1} as a function of the ratio of the solution height to the height at delayed critical h/h_c has been made in figure 9 for data of all fuel concentrations. These data are also tabulated in table IV. The calculated values of α_o^p are greater in absolute magnitude than the experimental values of H/X of 510, 975, and 1503 and are smaller at H/X of 1600. It is noted that the calculated experimental critical

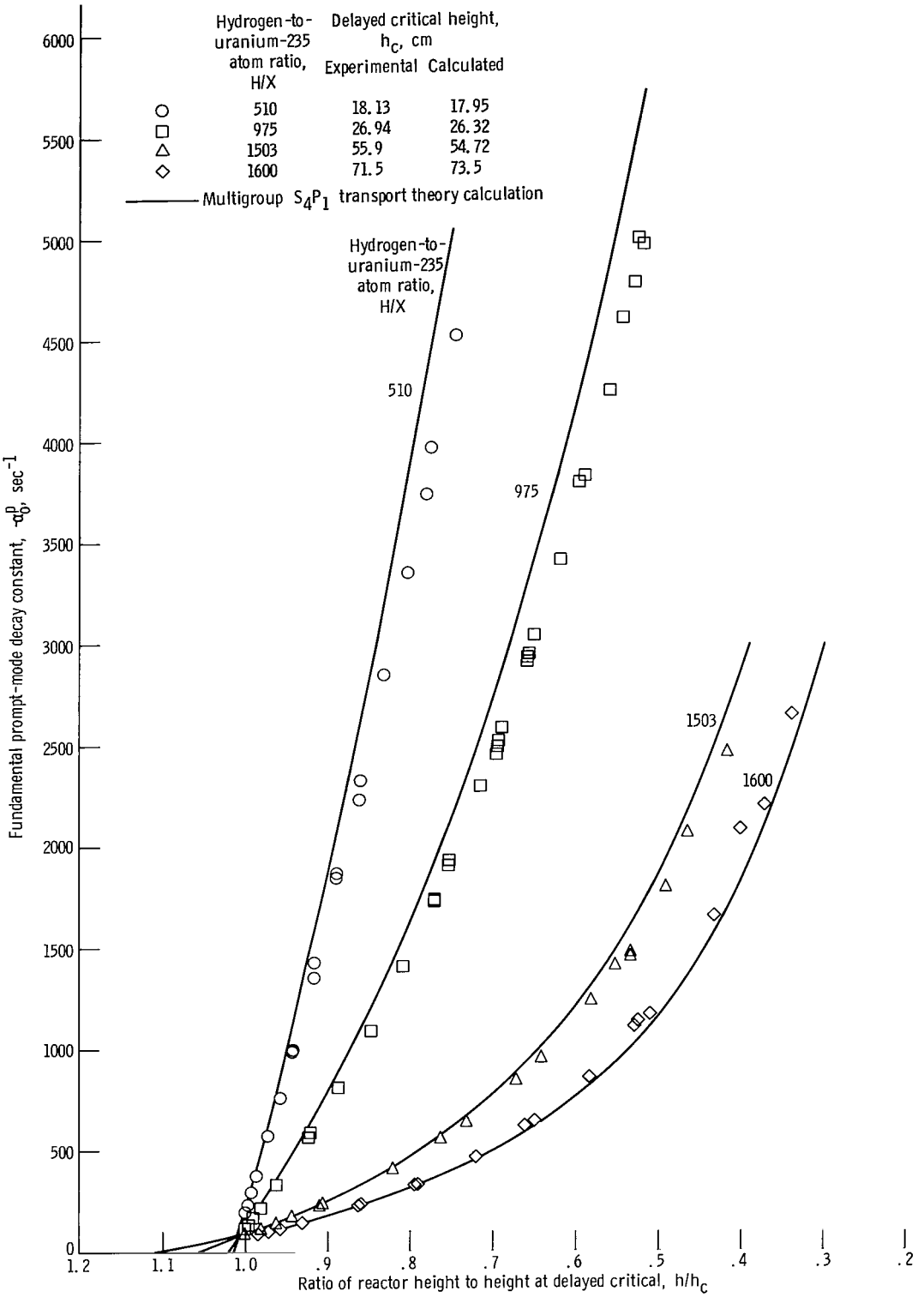


Figure 9. - Fundamental prompt-mode decay constants at various fuel concentrations as a function of reactor height parameter h/h_c.

TABLE IV. - TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF FUNDAMENTAL PROMPT-MODE DECAY CONSTANT

(a) At hydrogen-to-uranium-235 atom ratio H/X = 510

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}		Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}	
	Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$		Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$
[c]1.000	190±1.5	194	0.943	990±2.2	1125
[c]1.000	195±1.7	194	.918	1354±3.4	1565
[c].999	207±1.4	210	.917	1428±7	1580
[c].997	231±1.3	240	.890	1868±10	2075
[c].997	236±1.4	240	.889	1851±5.7	2095
[c].993	291±1.5	300	.864	2232±7.2	2570
[c].993	287±1.6	300	.861	2324±7.5	2620
[c].993	293±1.7	300	.833	2853±8.7	3220
[c].986	376±1.1	380	.805	3353±7.7	3860
[c].973	570±2.4	640	.783	3740±14	4400
.958	759±1.6	870	.777	3971±11	4530
[c].944	999±3.7	1110	.748	4524±16	5260

[a]Experimental h_c = 18.13 cm; calculated h_c = 17.95 cm.
[b]Experimental errors based on counting statistics only.
[c]He^3 detector; all others BF_3 detector.

heights are smaller at H/X of 510, 975, and 1503 but larger at H/X of 1600. The bias in the critical height calculation appears to vary inversely as might be expected when an h/h_c geometry parameter is used.

The experimental data points for α_o^p can be seen to follow a smooth curve at all concentrations and are in reasonable agreement with calculations. In all cases, the best agreement is found near delayed critical with the differences becoming increasingly larger for the farther shutdown cases. The differences between experiment and calculation varies with fuel concentration also. A direct comparison between experiments involving different fuel concentrations is always subject to question. However, some usable comparisons can be made.

The agreement between experiment and calculation can be compared in different ways. When a calculation is made of any given experiment, either the solution height or the α_o^p can be fixed in the calculation and an iterative solution of the other variable carried out. For example, if α_o^p is fixed at -2500, the difference between experimental and calculated solution heights are about 2, 3, 3.5, and 4.5 percent for H/X values of 510, 975, 1503, and 1600. On this basis the data at H/X of 510 agrees best. On the

TABLE IV. - Continued. TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF FUNDAMENTAL PROMPT MODE DECAY CONSTANT

(b) At hydrogen-to-uranium-235 atom ratio H/X = 975

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}		Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}	
	Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$		Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$
[c]1.000	115±0.6	116	0.755	1904±5	2105
[c]1.000	117±0.6	116	.753	1937±6	2130
[c].997	126±0.6	134	.717	2297±7	2549
[c].996	136±0.5	139	.696	2496±12	2800
[c].991	166±0.5	170	.695	2524±9	2815
[c].981	218±0.6	230	.691	2586±8	2860
.963	330±1	345	.660	2918±10	3290
[c].962	330±1	352	[c].660	2938±8	3290
[c].961	335±1	358	.658	2950±11	3320
.924	568±2	620	.652	3043±10	3400
[c].923	572±1	625	.622	3417±29	3880
[c].922	592±2	635	.599	3797±15	4240
[c].922	591±2	635	.592	3828±15	4360
.887	815±2	895	.592	3820±10	4360
.849	1096±4	1200	.563	4252±15	4900
.849	1092±3	1200	[c].547	4607±19	5220
[c,d].849	1091±4	1200	.546	4604±16	5240
[d].810	1404±6	1545	.533	4782±10	5500
.774	1732±5	1905	.527	5008±12	5650
.771	1749±6	1940	.522	4977±17	5760
[c].755	1914±7	2105			

[a]Experimental h_c = 26.94 cm; calculated h_c = 26.32 cm.

[b]Experimental errors based on counting statistics only.

[c]He^3 detector; all others BF_3 detector.

[d]Data differs from table III, ref. 11, which was in error.

TABLE IV. - Continued. TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF FUNDAMENTAL PROMPT-MODE DECAY CONSTANT

(c) At hydrogen-to-uranium-235 atom ratio H/X = 1503

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}		Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}	
	Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$		Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$
[c]1.000	79±1.5	80.5	0.764	551±2	570
[c]1.000	83±1.7	80.5	.732	637±2	670
[c]1.000	78±2.5	80.5	.673	840±2	885
[c]1.000	79±2.0	80.5	.643	954±3	1015
[c].982	108±0.5	108	.583	1233±3	1315
[c].963	134±0.5	138	.554	1404±4	1485
[c].944	164±0.5	171	[c].535	1450±4	1605
.911	221±0.7	230	[c].535	1466±4	1605
[c].907	229±0.5	238	.492	1793±6	1940
[c].907	228±0.5	238	.466	2060±5	2170
.821	400±1	420	.419	2458±7	2680

[a]Experimental h_c = 55.9 cm; calculated h_c = 54.72 cm.
[b]Experimental errors based on counting statistics only.
[c]He^3 detector; all others BF_3 detector.

other hand, if the α_o^p values are compared at a given fuel height in the same general range, the differences in α_o^p are about 10 percent for H/X of 510 and 975 and about 7 percent for H/X of 1503 and 1600. In this case the data at H/X of 510 give the poorest agreement and data at all fuel concentrations appear to give poorer agreement than before. This is another indication of the very sensitive nature of the parameter α_o^p which is the key parameter of these studies. The α_o^p values have been compared in reference 11 and again in this report because of the importance of this parameter even though its sensitive nature amplifies the difference between the experimental and calculated data. The comparisons of shutdown reactivities are also made for a given solution height in each case.

A measure of the reproducibility of the experimental α_o^p data can be found at H/X of 975 where several data points were repeated a number of times. The data from the other fuel concentrations (see tables IV(a), (c), and (d)) are limited. The data at H/X of 975 indicate that most of the data have a standard deviation of 1 percent of the value of α_o^p except for the most shutdown cases which increase to about 1.5 percent in standard deviation. At other concentrations where data points with the same h/h_c are repeated

TABLE IV. - Concluded. TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF FUNDAMENTAL PROMPT-MODE DECAY CONSTANT

(d) At hydrogen-to-uranium-235 atom ratio H/X = 1600

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}		Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Fundamental prompt-mode decay constant, sec^{-1}	
	Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$		Experimental[b] $-(\alpha_o^p)_E$	Calculated $-\alpha_o^p$
[c]1.000	80±0.7	77	0.791	338±0.8	325
[c]1.000	79±0.5	77	.721	472±2	455
[c].986	89±0.5	88	[c].662	623±2	590
[c].973	103±0.5	101	.650	648±2	620
[c].972	103±0.4	102	.584	863±3	820
[c].959	117±0.5	115	[c].530	1116±3	1025
[c].932	145±0.4	142	[c].525	1142±3	1045
[c].931	146±0.4	144	.511	1178±4	1115
[c].863	230±0.5	222	.434	1659±4	1565
.861	234±0.7	225	[c].402	2091±7	1820
.860	235±0.9	226	.373	2203±8	2100
[c].796	331±0.9	315	.340	2652±8	2480
[c].794	334±0.8	320			

[a]Experimental h_c = 71.5 cm; calculated h_c = 73.5 cm.
[b]Experimental errors based on counting statistics only.
[c]He^3 detector; all others BF_3 detector.

or nearly repeated, the reproducibility of α_o^p is of the order of 1 percent in those cases also. The exception to this degree of reproducibility occurs at delayed critical in the 510 and 1503 cases where the deviation is of the order of 2 to 2.5 percent. This is most likely due to the difficulties associated with pulsing a delayed critical system.

Determination of Shutdown Reactivity

Two general methods of determining shutdown reactivity from the experimental data have been used. One general method uses the values of α_o^p to determine reactivity and has two variations, an inhour technique and a technique of Simmons and King. The second general method uses the prompt- and delayed-neutron responses and has three variations, the "area-ratio" techniques of Sjöstrand, Garelis and Russell, and Gozani. Altogether five different determinations of shutdown reactivity worth are presented for each experimental measurement. All techniques are described briefly in the section

TABLE V. - TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF SHUTDOWN REACTIVITY

(a) At hydrogen-to-uranium-235 atom ratio H/X = 510

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Shutdown reactivity, dollars					
	Calculated $-\rho_s^C$	Experimental				
		Inhour[b] $-\rho_s^{IN}$	Simmons and King[b] $-\rho_s^{SK}$	Area-ratio methods[c]		
				Sjöstrand $-\rho_s^{SO}$	Garelis and Russell $-\rho_s^{GR}$	Gozani $-\rho_s^{GO}$
[d]0.999	0.07	0.05	0.07	0.11±0.0005	0.11±0.0005	0.11±0.0005
[d].997	.22	.18	.20	(e)	(e)	0.20±0.001
[d].993	.53	.48	.49	(e)	(e)	0.50±0.003
[d].986	.94	.92	.95	(e)	(e)	0.97±0.006
[d].973	2.29	1.93	1.95	1.90±0.02	1.98±0.02	1.98±0.02
.958	3.49	2.92	2.93	2.85±0.02	3.01±0.02	3.02±0.02
[d].944	4.77	4.19	4.18	3.82±0.07	4.09±0.08	4.11±0.08
.943	4.85	4.14	4.13	3.89±0.03	4.15±0.03	4.17±0.03
.918	7.22	6.11	6.02	5.61±0.07	6.13±0.07	6.19±0.07
.889	10.16	8.86	8.59	7.72±0.13	8.62±0.13	8.72±0.13
.864	12.86	11.04	10.56	10.00±0.22	11.00±0.21	11.12±0.22
.861	13.15	11.56	11.04	9.76±0.25	10.76±0.25	10.91±0.25
.833	16.66	14.65	13.78	12.14±0.32	13.58±0.31	13.83±0.32
.805	20.53	17.70	16.37	15.17±0.43	17.10±0.42	17.45±0.43
.783	23.89	20.15	18.38	17.13±0.43	19.48±0.42	19.94±0.43
.777	24.72	21.55	19.58	16.63±0.50	19.28±0.48	20.01±0.50
.748	29.44	25.18	22.44	21.07±0.68	24.27±0.66	24.97±0.67

[a]Experimental h_c = 18.13 cm; calculated h_c = 17.95 cm.

[b]Data obtained using $-\left(\alpha_o^p\right)_E$ values in table IV. Experimental errors can be inferred from same data.

[c]Experimental errors based on counting statistics only.

[d]He^3 detector; all others BF_3 detector.

[e]Result not obtainable from GRIPE II computer program.

PULSED-SOURCE EXPERIMENTS and in detail in reference 11. The data obtained using all five techniques are tabulated in table V along with calculated data. These data are also compared in figure 10.

Figure 10(a) is a plot of shutdown reactivity as a function of h/h_c comparing multigroup S_n transport calculations with both the inhour technique $\rho_s^{IN}(\$)$ and the technique of Simmons and King $\rho_s^{SK}(\$)$ for the four fuel concentrations studied. The solid lines represent the calculated values, while the dashed lines are used to show the experimental data points. The inhour technique gives significantly better agreement for all fuel concentrations studied except the H/X of 1600. The inhour technique uses the experimental α_o^p and calculated values of $\bar{\beta}_o/\Lambda_o$ given in figure 8 to determine shutdown reactivity. The Simmons and King technique determines the shutdown reactivity by using the measured values of α_o^p at the shutdown configuration in question and at delayed critical.

TABLE V. - Continued. TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF SHUTDOWN REACTIVITY

(b) At hydrogen-to-uranium-235 atom ratio H/X = 975

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Shutdown reactivity, dollars					
	Calculated $-\rho_s^C$	Experimental				
		Inhour[b] $-\rho_s^{IN}$	Simmons and King[b] $-\rho_s^{SK}$	Area-ratio methods[c]		
				Sjöstrand $-\rho_s^{SO}$	Garelis and Russell $-\rho_s^{GR}$	Gozani $-\rho_s^{GO}$
[d]0.997	0.17	0.10	0.09	0.14±0.0004	0.14±0.0004	0.14±0.0004
[d].996	.21	.19	.17	0.21±0.0006	0.21±0.0006	0.21±0.0006
[d].991	.48	.45	.43	0.44±0.002	0.46±0.002	0.46±0.002
[d].981	1.01	.91	.88	0.95±0.004	1.00±0.004	1.00±0.004
.963	2.03	1.90	1.84	1.81±0.01	1.90±0.01	1.90±0.01
.924	4.50	4.04	3.90	3.79±0.05	4.03±0.05	4.04±0.05
[d].922	4.63	4.25	4.10	3.96±0.05	4.23±0.05	4.22±0.05
.887	7.01	6.30	6.03	5.77±0.12	6.20±0.12	6.22±0.12
.849	9.86	8.92	8.45	7.73±0.14	8.55±0.14	8.64±0.14
.849	9.86	8.88	8.41	7.97±0.14	8.80±0.14	8.89±0.14
.810	13.16	11.87	11.10	10.19±0.22	11.49±0.22	11.69±0.22
.774	16.70	15.09	13.93	13.74±0.37	15.68±0.36	16.02±0.37
.771	17.04	15.27	14.08	13.66±0.35	15.40±0.35	15.61±0.35
.755	18.70	16.82	15.41	13.91±0.32	15.98±0.31	16.33±0.32
.753	18.95	17.15	15.70	13.94±0.35	16.17±0.34	16.70±0.35
.717	23.27	20.87	18.80	16.96±0.47	19.70±0.46	20.27±0.47
.696	25.92	23.00	20.52	19.39±0.60	22.75±0.58	23.34±0.60
.695	26.08	23.28	20.76	19.25±0.57	22.29±0.55	22.79±0.57
.691	26.57	23.93	21.29	19.96±0.68	23.59±0.65	24.50±0.68
.660	31.25	27.61	24.16	22.34±0.82	25.48±0.78	26.28±0.82
.658	31.58	27.95	24.43	23.05±0.87	26.35±0.83	27.20±0.87
.652	32.49	28.97	25.23	23.14±0.64	27.18±0.62	27.91±0.64
.622	37.96	33.31	28.46	25.83±0.92	31.01±0.89	32.12±0.92
.599	42.25	37.73	31.73	32.11±1.5	37.15±1.4	38.75±1.5
.592	43.70	38.25	32.00	27.83±1.5	32.30±1.4	33.79±1.5
.592	43.70	38.17	31.93	31.07±0.99	35.26±0.99	36.11±0.99
.563	50.42	43.62	35.66	34.22±1.5	39.38±1.4	40.62±1.5
.546	54.80	48.03	38.69	37.25±1.3	43.51±1.2	45.37±1.3
.533	58.31	50.57	40.22	41.54±1.4	48.34±1.3	50.00±1.4
.527	60.15	53.20	42.17	41.50±2.2	49.54±2.1	52.64±2.2
.522	61.75	53.22	41.91	43.09±2.4	50.11±2.3	51.84±2.4

[a]Experimental h_c = 26.94 cm; calculated h_c = 26.32 cm.

[b]Data obtained using $-\left(\alpha^p_o\right)_E$ values in table IV. Experimental errors can be inferred from same data.

[c]Experimental errors based on counting statistics only.

[d]He^3 detector; all others BF_3 detector.

TABLE V. - Continued. TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF SHUTDOWN REACTIVITY

(c) At hydrogen-to-uranium-235 atom ratio H/X = 1503

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Shutdown reactivity, dollars					
	Calculated $-\rho_s^C$	Experimental				
		Inhour[b] $-\rho_s^{IN}$	Simmons and King[b] $-\rho_s^{SK}$	Area-ratio methods[c]		
				Sjöstrand $-\rho_s^{SO}$	Garelis and Russell $-\rho_s^{GR}$	Gozani $-\rho_s^{GO}$
[d]0.982	0.33	0.33	0.35	(e)	(e)	0.31±0.001
[d].963	.71	.66	.68	(e)	0.65±0.003	0.65±0.003
[d].944	1.12	1.03	1.05	(e)	1.02±0.005	1.02±0.005
.911	1.85	1.74	1.76	1.60±0.01	1.73±0.01	1.73±0.01
[d].907	1.95	1.84	1.86	1.60±0.008	1.75±0.008	1.76±0.008
[d].907	1.95	1.83	1.85	1.61±0.007	1.75±0.007	1.76±0.007
.821	4.25	4.00	4.00	3.58±0.04	4.00±0.04	4.04±0.04
.764	6.17	5.93	5.89	4.96±0.06	5.96±0.06	5.79±0.06
.732	7.46	7.04	6.96	6.01±0.11	6.84±0.11	6.97±0.11
.673	10.27	9.70	9.50	8.17±0.19	9.46±0.18	9.70±0.19
.643	12.00	11.22	10.93	9.06±0.22	10.43±0.22	10.67±0.22
.583	16.06	15.00	14.41	12.03±0.17	14.12±0.16	14.61±0.17
.554	18.42	17.36	16.55	13.94±0.33	16.41±0.32	17.00±0.33
[d].535	20.10	18.07	17.13	13.83±0.31	16.44±0.30	17.12±0.31
[d].535	20.10	18.28	17.33	14.49±0.32	17.22±0.30	17.96±0.32
.492	24.90	22.94	21.41	18.50±0.36	21.85±0.34	22.65±0.36
.466	28.28	26.80	24.75	20.98±0.63	24.45±0.59	25.58±0.63
.419	36.07	33.00	29.73	26.80±1.00	31.20±0.95	32.65±1.00

[a]Experimental h_c = 55.9 cm; calculated h_c = 54.72 cm.

[b]Data obtained using $-\left(\alpha^p_o\right)_E$ values in table IV. Experimental errors can be inferred from same data.

[c]Experimental errors based on counting statistics only.

[d]He^3 detector; all others BF_3 detector.

[e]Result not obtainable from GRIPE II computer program.

TABLE V. - Concluded. TABULATION OF EXPERIMENTAL AND CALCULATED VALUES OF SHUTDOWN REACTIVITY

(d) At hydrogen-to-uranium-235 atom ratio H/X = 1600

Ratio of fuel solution height to fuel height at delayed critical,[a] h/h_c	Shutdown reactivity, dollars					
	Calculated $-\rho_s^C$	Experimental				
		Inhour[b] $-\rho_s^{IN}$	Simmons and King[b] $-\rho_s^{SK}$	Area-ratio methods[c]		
				Sjöstrand $-\rho_s^{SO}$	Garelis and Russell $-\rho_s^{GR}$	Gozani $-\rho_s^{GO}$
[d]0.986	0.14	0.15	0.12	(e)	(e)	0.16±0.00005
[d].973	.31	.34	.30	0.29±0.001	0.32±0.002	0.32±0.001
[d].972	.32	.34	.30	(e)	0.33±0.001	0.33±0.0009
[d].959	.49	.52	.47	0.45±0.002	0.49±0.002	0.49±0.002
[d].932	.85	.88	.82	0.77±0.003	0.85±0.003	0.85±0.003
[d].931	.87	.90	.84	(e)	0.87±0.002	0.88±0.002
[d].863	1.89	2.00	1.89	1.69±0.008	1.91±0.008	1.93±0.008
.861	1.93	2.05	1.94	1.73±0.009	1.98±0.01	2.00±0.009
.860	1.95	2.06	1.96	1.77±0.009	2.01±0.01	2.02±0.009
[d].796	3.12	3.33	3.16	2.72±0.03	3.14±0.03	3.19±0.03
[d].794	3.19	3.37	3.20	2.86±0.02	3.31±0.02	3.37±0.02
.791	3.25	3.42	3.25	2.88±0.02	3.34±0.02	3.40±0.02
.721	4.99	5.21	4.94	4.30±0.05	5.08±0.05	5.22±0.05
[d].662	6.80	7.24	6.84	5.70±0.07	6.79±0.07	7.03±0.07
.650	7.21	7.58	7.15	6.25±0.09	7.52±0.09	7.79±0.09
.584	9.96	10.53	9.86	8.36±0.13	10.19±0.12	10.65±0.13
[d].530	12.82	14.05	13.04	10.69±0.25	12.87±0.24	13.46±0.25
[d].525	13.11	14.42	13.36	11.24±0.20	13.61±0.19	14.29±0.20
.511	14.10	14.95	13.82	11.62±0.30	14.12±0.29	14.85±0.30
.434	20.63	21.93	19.87	17.40±0.48	21.04±0.45	22.13±0.48
[d].402	24.47	28.27	25.30	21.93±0.78	27.12±0.72	28.99±0.78
.373	28.77	30.23	26.71	23.90±0.83	29.29±0.76	31.22±0.83
.340	34.79	37.27	32.36	29.58±1.17	35.00±1.09	37.05±1.17

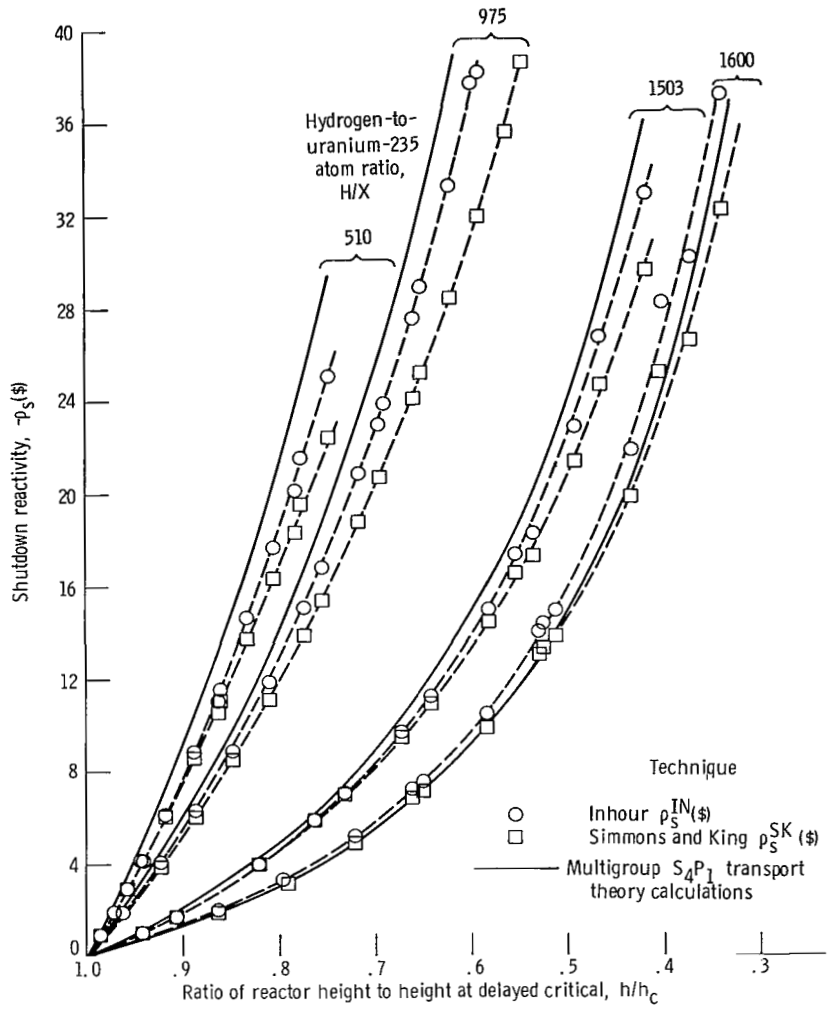
[a]Experimental h_c = 71.5 cm; calculated h_c = 73.5 cm.

[b]Data obtained using $-\left(\alpha_o^p\right)_E$ values in table IV. Experimental errors can be inferred from same data.

[c]Experimental errors based on counting statistics only.

[d]He^3 detector; all others BF_3 detector.

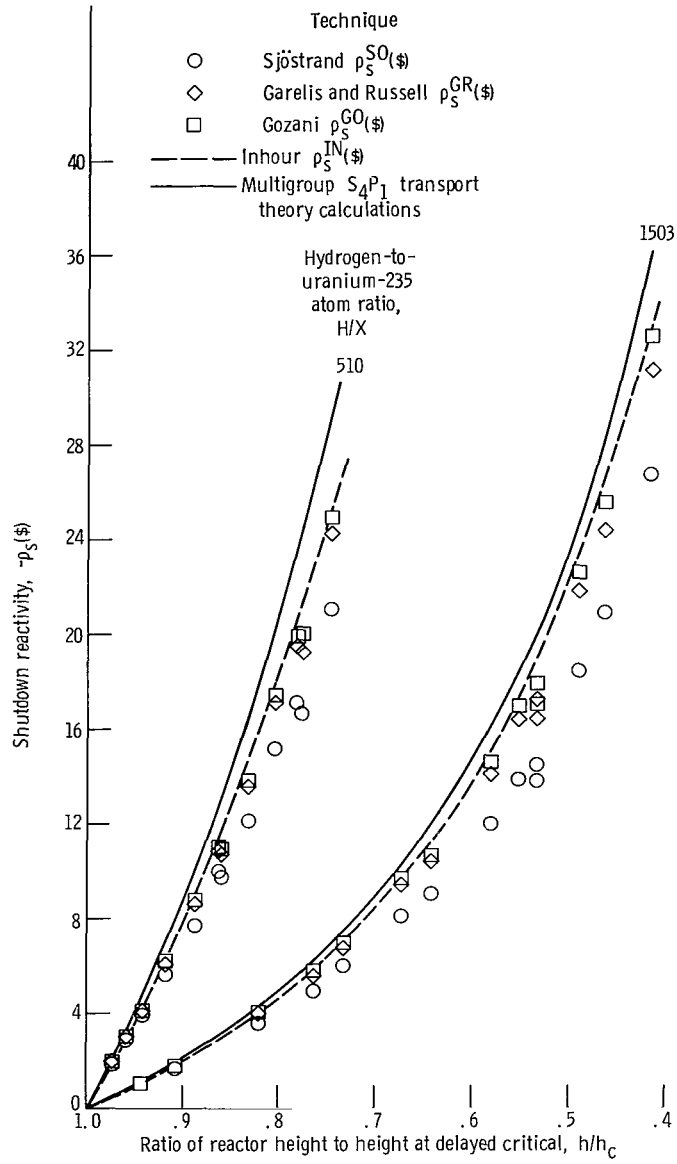
[e]Result not obtainable from GRIPE II computer program.



(a) Simmons and King and inhour techniques.

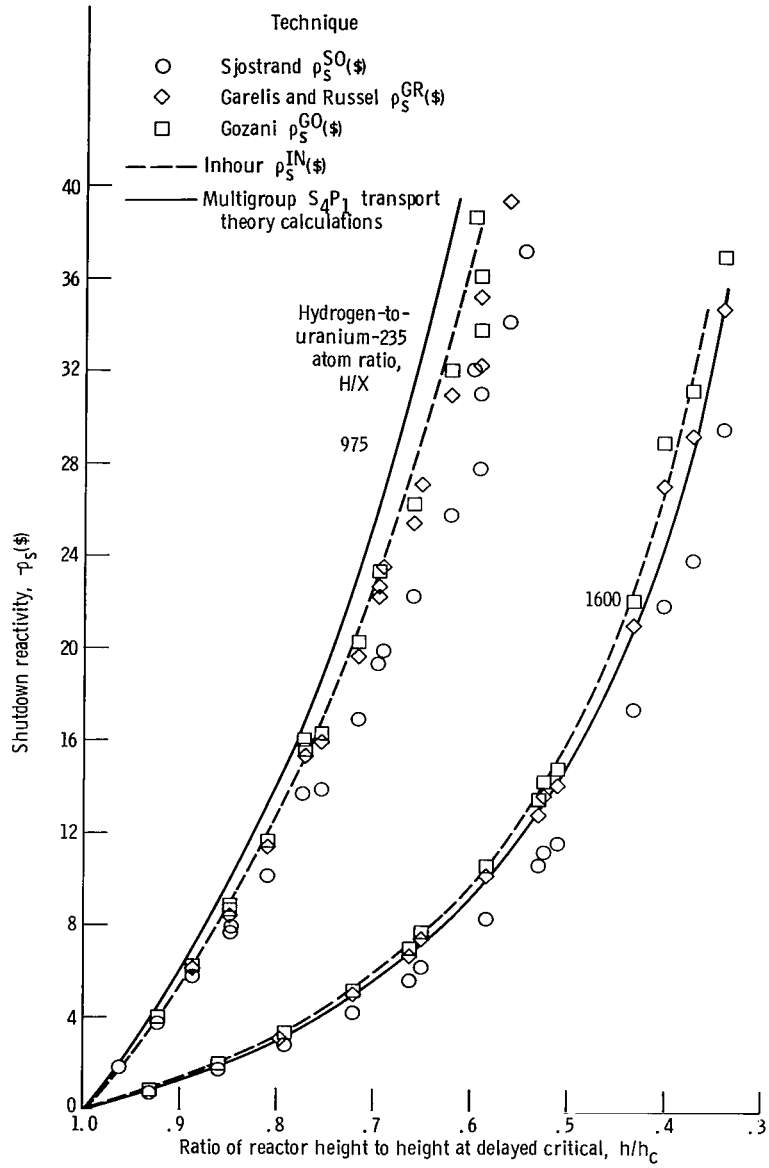
Figure 10. - Comparison of experimental and calculated shutdown reactivities at various fuel concentrations as a function of reactor height parameter h/h_c. (See table III for solution heights at delayed critical.)

The fact that the calculated parameter $\overline{\beta}_o/\Lambda_o$ (which is $-\alpha_o^p$ at delayed critical) does not vary nearly so much at H/X of 1503 and 1600 as it does for H/X of 510 and 975 is at least part of the reason why the Simmons and King technique gives better agreement at the less concentrated solutions than at the more concentrated solutions. This probably does not completely explain the excellent agreement of the reactivities obtained using the Simmons and King technique with transport calculations at H/X of 1600. In any case, it must be concluded that the assumption that $\overline{\beta}_o/\Lambda_o$ is a constant for a particular fuel concentration is not generally valid and the Simmons and King technique has definite limitations for this reason.



(b) Area-ratio techniques at H/X of 510 and 1503.

Figure 10. - Continued.



(c) Area-ratio techniques at H/X of 975 and 1600.

Figure 10. - Concluded.

The shutdown reactivities determined by the "area" techniques are compared with the transport calculations and the inhour data curve in figures 10(b) and (c). The data from each of the three "area" techniques show a larger scatter than the data for $\rho_s(\$)$ previously presented. This scatter is due primarily to the statistical error in the measurement of the delayed-neutron "area". In general, the shutdown reactivity data obtained by using the technique of Gozani $\rho_s^{GO}(\$)$ are in very good agreement with the data obtained by using the inhour technique. Both sets of data give the best agreement with the transport theory calculations over the range of concentrations considered in this report. The shutdown reactivity data obtained using the technique of Garelis and Russell $\rho_s^{GR}(\$)$ gives only slightly poorer overall agreement with calculations. The experimental reactivity obtained using the technique of Sjöstrand $\rho_s^{SO}(\$)$ gave the poorest agreement of the five techniques in most instances.

The agreement between the calculated reactivity and the experimentally determined data was poorest at H/X of 510 and became better with a decrease in fuel concentration. Comparing experimental inhour technique data with calculated data (see fig. 10(a)) at about \$30 shutdown for each concentration, the difference is about 18 percent at H/X of 510, 12 percent at 975, 8 percent at 1503, and 8 percent at 1600. At \$10 shutdown the differences are about 13, 8, 6, and 6 percent, respectively. Comparison has been made in all cases using the calculated value as the reference. The reactivity results obtained using the Gozani technique were essentially the same as the results from the inhour technique except for a larger scatter.

Comparing calculated values with experimental values obtained using Simmons and King technique at \$30 shutdown yields differences of about 24 percent at H/X of 510, 21 percent at 975, 16 percent at 1503, and 4 percent at 1600 (see fig. 10(a)). At \$10 shutdown the values are about 16, 13, 8, and <1 percent, respectively. The experimental reactivities obtained with the Sjöstrand technique agree fairly well with the Simmons and King results but are in poorer agreement with the calculated results at all concentrations.

The reactivities obtained using the Garelis and Russell technique give somewhat poorer agreement with calculation than either of the reactivities obtained using the inhour or Gozani techniques for all concentrations except at H/X of 1600. The transport theory calculations at H/X of 1600 generate a curve that makes a good average of the data from all five techniques for obtaining the experimental reactivity and within a few percent agreement with the Garelis and Russell data.

CONCLUSIONS

A number of conclusions result from this study of unreflected, homogeneous, thermal

reactors covering a wide range of fuel solution concentrations and from delayed critical to $25 or more shutdown; they are as follows:

1. In order to get one-dimensional multigroup-transport-theory-calculated values of the fundamental prompt-mode decay constants to more closely agree with experiment over the entire range of fuel concentration, it has been necessary to use a calculational method which closely predicts the measured height at delayed critical for all concentrations. This was accomplished by using a buckling iteration technique. Two-dimensional calculations were not used because a suitable two-dimensional transport program was not available.

2. Experimentally determined reactivities based on an inhour technique which uses the experimental value for the fundamental prompt-mode decay constant and a calculated parameter agree closely with experimental values obtained using the Gozani "area-ratio" technique. This inhour technique gives the most consistent results and provides the best overall agreement with the calculated reactivity values for the entire range of fuel concentrations.

3. The technique of Simmons and King gave experimental values for the reactivity which differed considerably from the calculated values for all fuel concentrations except for the hydrogen-to-uranium-235 atom ratio H/X of 1600. The better agreement at H/X of 1600 is primarily attributed to the fact that the ratio of the effective delayed-neutron fraction to the generation time remains nearly constant over the range of shutdown reactivities measured at this concentration. In general, this ratio varies significantly causing poor agreement with calculation for highly shutdown cases.

4. The Garelis and Russell "area-ratio" technique predicted experimental values for reactivity slightly lower than the values obtained using the Gozani area-ratio technique in all cases. Consequently, the values obtained using the Garelis and Russell technique are in slightly poorer agreement with calculation than the values obtained using the Gozani technique at H/X values of 510, 975, and 1503 but in slightly better agreement at H/X of 1600.

5. The "area-ratio" technique of Sjöstrand for determining experimental reactivity gave the poorest agreement with calculated values over the range of measurements and concentrations.

6. All of the area-ratio techniques for determining reactivity are sensitive to an accurate experimental determination of the equilibrium delayed-neutron background. Experimental reactivities obtained by using any of the area-ratio techniques show a greater amount of scatter for this reason.

Each of the five techniques for determining shutdown reactivity has advantages and disadvantages. The inhour technique and the Gozani and Garelis and Russell area-ratio techniques give comparable results in reasonably good agreement with transport calculations. The inhour method uses an experimental measurement of the fundamental

prompt-mode decay constant and a calculated parameter, $\overline{\beta}_o/\Lambda_o$ which is the ratio of the effective delayed-neutron fraction to the generation time for the prompt mode, in order to get reactivity. This calculated value is relatively easy to obtain and its value at delayed critical can be easily verified experimentally if desired. All area-ratio techniques require only a single experimental measurement but are subject to kinetic distortion factors and are sensitive to an accurate determination of the delayed-neutron background both of which can be difficult to ascertain. The technique of Simmons and King is direct and simple but requires the measurement of two different constants. This technique is also quite limited in range of use. The use of the inhour technique therefore is the most advantageous for this experimental arrangement.

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Cleveland, Ohio, May 19, 1970,
120-27.

REFERENCES

1. Sjöstrand, Nils G.: Measurements on a Subcritical Reactor Using a Pulsed Neutron Source. Arkiv Fysik, vol. 11, 1956, pp. 233-246.

2. Gozani, Tsahi: A Modified Procedure for the Evaluation of Pulsed Source Experiments in Subcritical Reactors. Nukleonik, vol. 4, no. 8, 1962, pp. 348-349.

3. Garelis, Edward; and Russell, John L., Jr.: Theory of Pulsed Neutron Source Measurements. Nucl. Sci. Eng., vol. 16, no. 3, July 1963, pp. 263-270.

4. Preskitt, C. A.; Nephew, E. A.; Brown, J. R.; and Van Howe, K. R.: Interpretation of Pulsed-Source Experiments in the Peach Bottom HTGR. Nucl. Sci. Eng., vol. 29, no. 2, Aug. 1967, pp. 283-295.

5. Simmons, B. E.; and King, J. S.: A Pulsed Neutron Technique for Reactivity Determination. Nucl. Sci. Eng., vol. 3, no. 5, May 1958, pp. 595-608.

6. Corngold, Noel: On the Analysis of Pulsed, Multiplying Systems. Rep. GA-5404, General Dynamics Corp., June 24, 1964.

7. Becker, Martin; and Quisenberry, Karl S.: The Spatial Dependence of Pulsed-Neutron Reactivity Measurements. Symposium on Neutron Dynamics and Control, Univ. of Ariz., 1965.

8. Masters, Christopher F.; and Cady, K. B.: A Procedure for Evaluating Modified Pulsed-Neutron-Source Experiments in Subcritical Nuclear Reactors. Nucl. Sci. Eng., vol. 29, no. 2, Aug. 1967, pp. 272-282.

9. Wallace, S. K.; Teare, K. R.; and Green, J. B.: Methods for the Comparison of Pulsed-Neutron Shutdown Measurements with Theory. Nucl. Sci. Eng., vol. 25, no. 4, Aug. 1966, pp. 407-412.

10. Fox, Thomas A.; Mueller, Robert A.; Ford, C. Hubbard; and Alger, Donald L.: Critical Mass Studies with NASA Zero Power Reactor II. I: Clean Homogeneous Configurations. NASA TN D-3097, 1965.

11. Fieno, Daniel; Fox, Thomas A.; Mueller, Robert A.; and Ford, C. Hubbard: Analysis and Experiments with a Pulsed Neutron Source for an Unreflected Solution Reactor up to $50 Shutdown. NASA TN D-5487, 1969.

12. Kaufman, N. C.: GRIPE II - A Computer Program for the Analysis of Data from a Pulsed-Neutron Experiment. Rep. IN-1085, Idaho Falls National Reactor Testing Station, Sept. 1967.

13. Joanou, G. D.; and Dudek, J. S.: GAM-II. A B_3 Code for the Calculation of Fast-Neutron Spectra and Associated Multigroup Constants. Rep. GA-4265, General Dynamics Corp., Sept. 16, 1963.

14. Joanou, G. D.; Smith, C. V.; and Vieweg, H. A.: GATHER-II. An IBM-7090 FORTRAN-II Program for the Computation of Thermal-Neutron Spectra and Associated Multigroup Cross Sections. Rep. GA-4132, General Dynamics Corp., July 8, 1963.

15. Keepin, G. Robert: Physics of Nuclear Kinetics. Addison-Wesley Publ. Co., 1965.

16. Alger, Donald; Mayo, Wendell; and Mueller, Robert: Measurement of Effective Delayed Neutron Fraction for NASA Zero Power Reactor I. NASA TN D-3709, 1966.

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